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FLIBE/Be/He/FS CONCEPT (R=1)

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8.7 Flibe/Be/He/FS Concept (R=1)

8.7.1 Design Choices

The blanket concept is shown in Fig. 8.7-1. Beryllium, in the form of pebbles nominally 1 cm diameter in a 20 cm thick bed, is employed to multiply neutrons. The multiplier zone is followed by a zone of silicon carbide (SiC) which slows neutrons. Neutrons are captured in the Lithium-6 carried in the molten fluoride salt (LiF+BeF- melting point, 363°C) to breed tritium and release extra energy in exothermic nuclear reactions. The salt flows slowly through tubes in the blanket and out to a simple flash separator where the tritium is removed. Helium flows radially through the Be pebble bed and SiC region carrying the heat out to the thermal conversion plant. The tubes are coated either on the inside or outside with a 10 μm tungsten barrier by chemical vapor deposition to cut down tritium permeation to the helium coolant circuit. With the tungsten barrier on the inside the tritium inventory in the tube walls is small and tungsten will contribute to corrosion inhibition. A 1-mm aluminum jacket on the steam generator tubes keeps the tritium permeation to the steam down to 30 curies per day. The design can be converted into a fission-suppressed fissile breeder by thickening the beryllium zone by a factor of 2 or so and adding ThF_4 to the salt in which case 6 tonnes of uranium - 233 would be produced per year.

Beryllium is chosen as the neutron multiplier because of its large (n,2n) nuclear reaction cross-section and low cross-section for competing side reactions. By comparison to other materials, beryllium significantly stands out as a neutron multiplier as can be seen in the infinite media results of Fig. 8.7-2. The material having the next largest neutron multiplication* is ^{7}Li. However, because its density is 3.8 times lower than beryllium, it requires a very thick blanket to approach its infinite media multiplication ability. The multiplication in Pb is much lower than in beryllium but is still quite appreciable.

Beryllium is a limited resource and some people have thought it inadvisable to use in fusion plant designs because then fusion power would

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*^{7}Li is not truly a multiplier but has the same effect because of the ^{7}Li(n,n'T)^{4}He reaction which produces tritium and preserves the incident neutron.

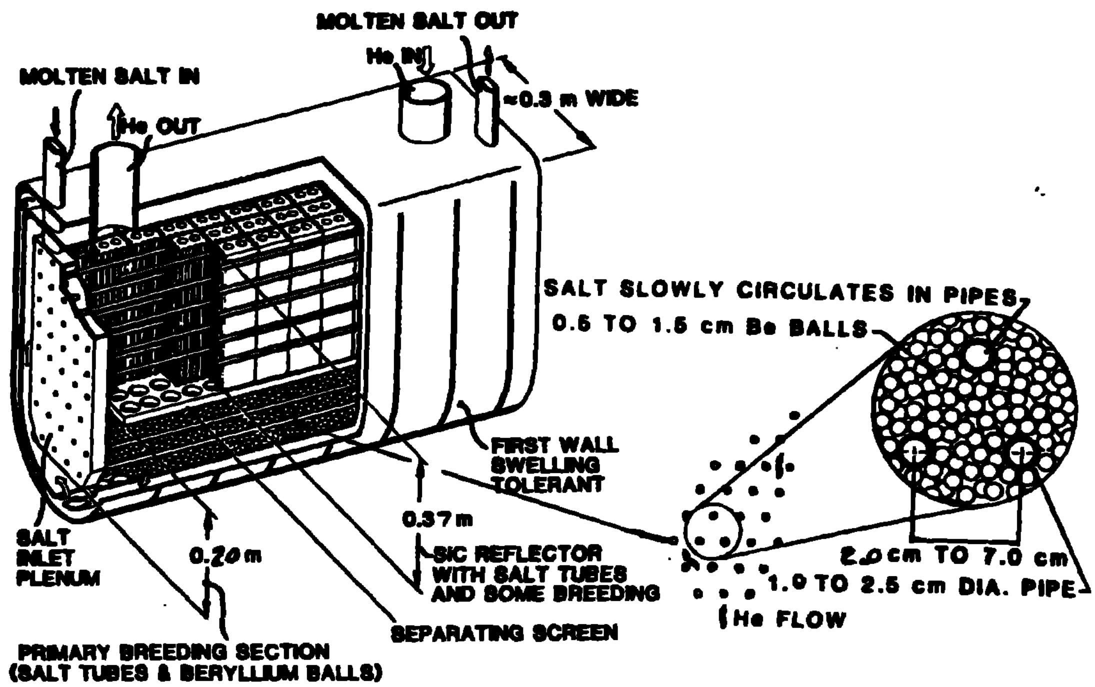


Figure 8.7-1
Helium cooled, flibe blanket configuration
with beryllium as the neutron multiplier.

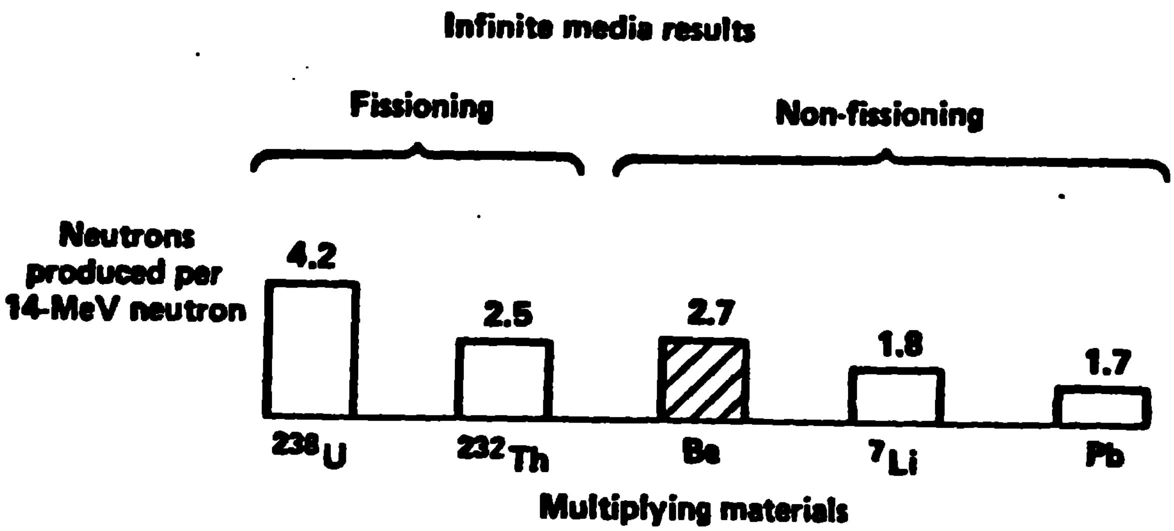


Figure 8.7-2
Neutron multiplication for various materials.
We chose beryllium because its multiplication was the highest
for nonfissioning materials.

not be inexhaustible. We find beryllium so advantageous that this question needs re-examining carefully. There is enough beryllium for many hundreds of plants and this would allow fusion to be deployed extensively enough for people to become familiar with the technology. After the first 50-70 years of introduction, the amount of beryllium employed in the blanket could be reduced by more neutronically efficient designs. Apparently, with careful design and full recycle of used beryllium, fusion power based on the use of beryllium can be considered semi-inexhaustible. Beryllium resources were discussed in last years report and elsewhere in this report.

We chose beryllium in the form of pebbles to facilitate recycle of irradiated beryllium. A bed of pebbles can potentially accommodate some swelling and relative thermal expansion, and can be loaded and unloaded by flowing. We envision blankets to be factory made, shipped to the plant and installed after testing and inspection. The recycled beryllium would have a contact dose rate that would not allow extensive personnel exposure. Therefore, the pebbles would be loaded into the blanket sometime after manufacturing, perhaps at the plant. If beryllium in a form not suitable for flowing were used, then the beryllium, or any other recycled material, would have to be loaded into the blanket during the manufacturing process and would require remote manufacturing of blankets. No one has shown how to fabricate something as complicated as a blanket by remote methods. Surely it can be done, but the cost may be prohibitive, not just double the usual manufacturing cost, for example.

Flibe was chosen as the tritium breeding material. It has been shown to have a very low corrosion rate with austenitic steel and is expected to be low with ferritic steel if the salt is kept in a reducing state (deficiency of flourine). Flibe is one of the few lithium bearing materials that will not react exothermically with air or water. Therefore, catastrophic accidents caused by such reactions are not possible.

The activation products for both flibe and Beryllium have a relatively short half-life so that they can be disposed of by shallow burial. After ten years, out-of-the-reactor worker protection is still needed; however, after 100 years flibe and beryllium can be handled without worker protection. Flibe has an extremely low tritium solubility which makes for very easy tritium removal but also leads to an increased tendency for tritium to permeate into the helium stream.

The blanket configuration shown in Fig. 8.7-1 was chosen for this study. This basic pod design for the helium pressure vessel forms the basis for most of the helium cooled designs and therefore, the first wall, manifolding, plena, and many other features are in common and do not come up as a unique issue when considering the flibe design.

We prefer to be able to gravity drain both the flibe and the beryllium pebbles. Both these objectives were achieved in the design developed for the tandem mirror. However, due to the non-commonality with the other helium cooled designs considered in the BCSS, we relegated this to a "backup" design. It is shown in Fig. 8.7-3. The end view is shown in Fig. 8.7-4. An adaptation to the tokamak configuration has been worked out but is not shown.

8.7.2 Tokamak Blanket Configuration

The pod modules shown in Fig. 8.7-1 are arranged to fit the Tokamak configuration as shown in Fig. 8.3-1. The module ends, where one sector fits very close to the next sector, is not shown, but is discussed elsewhere in this report. Much more effort needs to be devoted to module end design as it has a major impact on blanket design and is not not just a perturbation to the design.

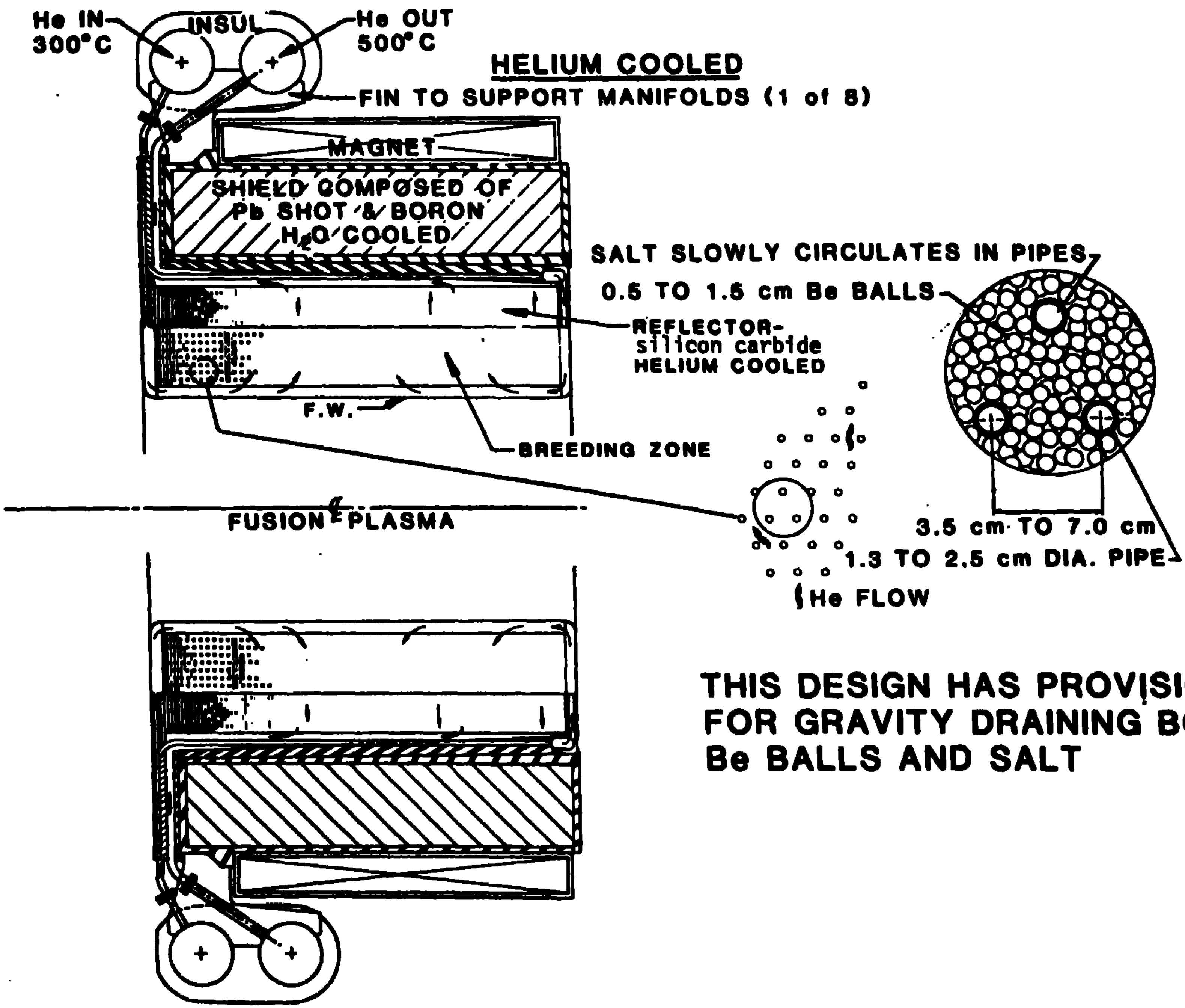
8.7.3 Tandem Mirror Blanket Configuration

The pod modules shown in Fig. 8.7-1 are arranged to fit the Tandem Mirror configuration as shown in Fig. 8.3-2. The dimensions are based on the MARS study (B.G. Logan et. al., "Mirror Advanced Reactor Study (MARS)", Lawrence Livermore National Laboratory Report UCRL-53563, (1984)). The first wall radius is 0.6 m and the center cell (blankets) is 130 m long. The vacuum magnetic field on axis is 4.7 T.

8.7.4 Design Summary and Issues

The key issue with the Flibe design is tritium control. The tungsten on the flibe tubes and the aluminum jacket including the oxide layer on the steam generator tubes must be shown to be workable and reliable as tritium barriers.

Another important issue is integrity of beryllium during its residence in the blanket. A limited amount of breakup could be tolerated to the point where flying particles would damage the helium circulator or plug the pebble bed. We speculate a two year residence time (8 MWy/m^2) might be economically



THIS DESIGN HAS PROVISION FOR GRAVITY DRAINING BOTH Be BALLS AND SALT

Figure 8.7-3

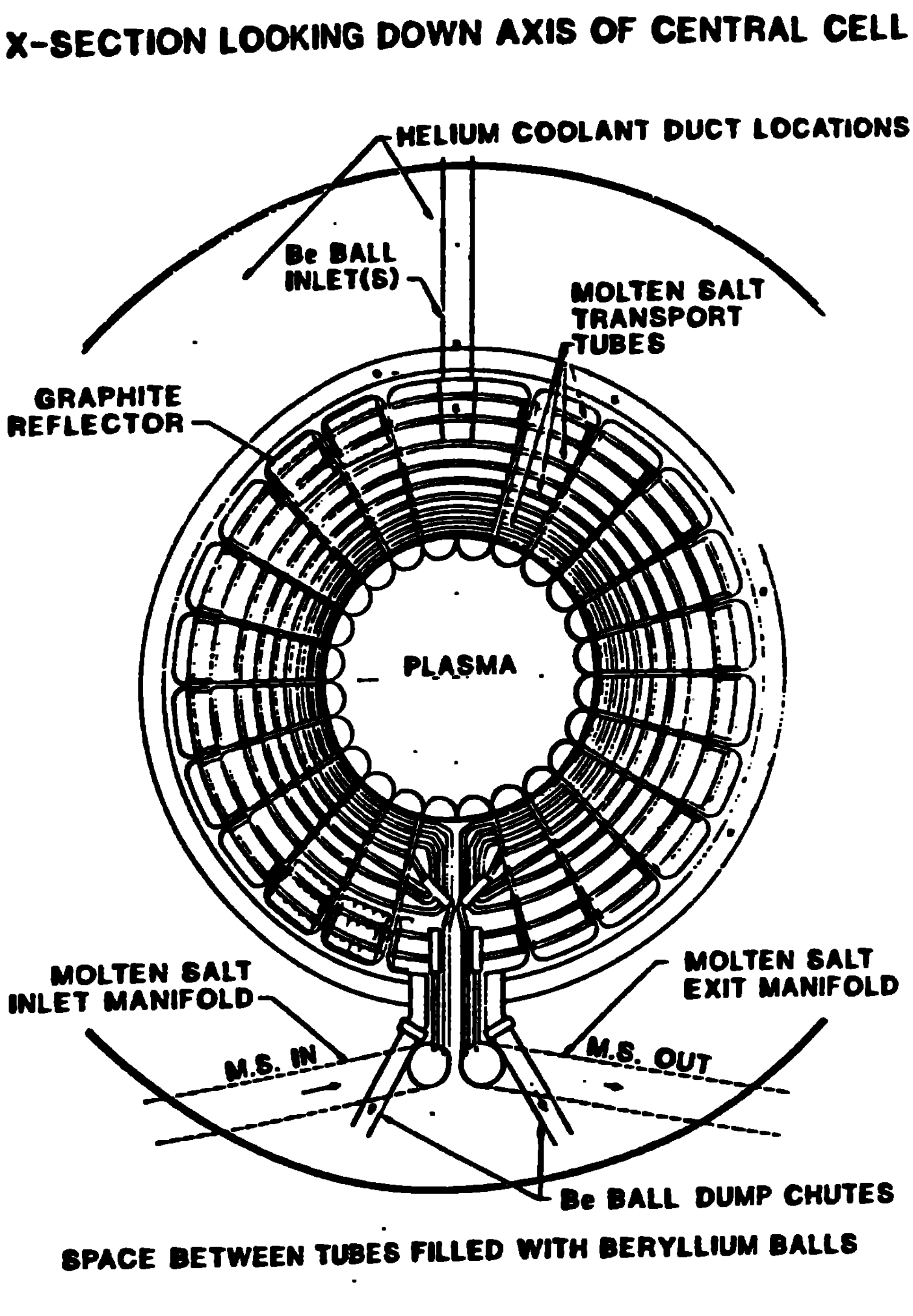


Figure 8.7-4

Molten salt blanket.

acceptable after which time the pebbles could be remanufactured especially with the back-up design in which the pebbles could be removed and replaced on a shorter time than the blanket changeout time.

The question of tritium which is generated in the beryllium being released into the helium was not addressed. Since only 1% of the tritium is produced in the beryllium, the problem was considered unimportant.

Some people believe the remanufacturing of beryllium pebbles will require a large development effort. We believe, however, that use of automated, free flowing powder techniques now being implemented in the beryllium industry will allow remanufacturing of these pebbles by straight forward but automated powder metallurgical techniques without a large developmental effort.

Hydrogen (tritium) embrittlement has been flagged as a special problem for the flibe design because the tritium concentration estimated for the HT-9 tubes is 1.5 wppm versus 0.3 to 0.6 wppm for the other designs not using flibe. This 1.5 wppm was appropriate for the tungsten layer on the outside of the tubes. With tungsten on the inside the tritium concentration should go down by a large factor. Experiments will be needed to determine under realistic operating conditions the actual tritium concentration and see if this is any problem whatsoever.

Corrosion of HT-9 tubes by flibe has been flagged as a significant problem for the flibe design by the BCSS. With austenitic steel the corrosion has been shown experimentally to be very small, and with ferritic steel we expect it to be very low. Pumped loop experiments will be needed to prove the corrosion rates are low, especially with realistic impurities. MHD and radiation effects are not predicted to be important.